<u>ENCLOSURE 2</u>

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
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Dates:	October 26-31, 1997, with in-office inspection continuing through January 20, 1998
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Attachment 1	Supplemental Information
Attachment 2	Licensee-Provided Supplemental Information

EXECUTIVE SUMMARY

River Bend Station
NRC Inspection Report 50-458/97-09

River Bend Station personnel developed and implemented a program in accordance with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," with a few exceptions noted. The team found that: the performance or condition of six structures, systems, and/or components (SSCs) had not been demonstrated to have been effectively controlled through the performance of appropriate preventive maintenance; the governing procedures and guidelines for implementing the Maintenance Rule program were weak as evidenced by a lack of clear boundary definitions, function definitions, and quantitative performance measures. The team also found that the online and shutdown maintenance risk-assessment programs were well developed and implemented, and that the persons interviewed were knowledgeable of the risk-assessment process.

Supplemental information was provided to the team on November 18 and 20, 1997. This information is included as Attachment 2.

Maintenance

- Scoping of SSCs for inclusion in the Maintenance Rule program was generally acceptable. A noncited violation was identified for the failure to include six SSCs within the scope of the Maintenance Rule program on July 10, 1996 (Section M1.1).

- The risk-significance ranking process was generally comprehensive (Section M1.2).

- The licensee's use of the conditional probability [of success] had the potential to mask adverse trends in system performance because of the combination of the availability and reliability measures into one measure (Section M1.2).

- Reliability and availability performance measures for the risk-significant, normally operating SSCs were adequate to balance availability and reliability (Section M1.4).

- The licensee's online and shutdown maintenance risk-assessment programs were well developed and implemented, with use of the equipment out-of-service computer program for both online and shutdown risk-assessment. The persons interviewed expressed good knowledge of the risk-assessment process and a strong sense of involvement and ownership (Section M1.5).

- Four examples of a violation of 10 CFR 50.65(a)(2) were identified for the failure to demonstrate that the performance or condition of SSCs had been effectively controlled through the performance of appropriate preventive maintenance (Section M1.6).

- The governing procedures and guidelines for implementing the Maintenance Rule program were weak, as evidenced by a lack of clear boundary definitions, function definitions, and quantitative performance measures (Section M3).

Engineering

- The level of experience of the expert panel members and the frequency of the expert panel meetings were strengths in the implementation of the Maintenance Rule program (Section E4).

- The system engineers' knowledge was generally consistent with their Maintenance Rule program responsibilities (Section E4).

Plant Support

- Housekeeping was very good, especially considering that the plant had been recently returned to operation after a refueling outage (Section M2).

DETAILS

Summary of Plant Status

The plant was operating at 100 percent power during this inspection.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Scope of the System, Structure, and Component Functions Included Within the Maintenance Rule

a. Inspection Scope (62706)

The team reviewed the licensee's procedure for initial scoping, the River Bend Station Updated Final Safety Analysis Report, and emergency operating procedures. The team reviewed the list of structures, systems, and components (SSCs) that the licensee had included in the scope of the Maintenance Rule program, and developed an independent list of SSCs that were not included by the licensee. The team used this list to determine if the licensee had adequately identified the SSC functions that should have been included in the scope.

b. Observations and Findings

The team noted that six nonsafety-related SSCs had been added to the scope recently by the expert panel. The communications system; the emergency lighting system; the turbine building; the radioactive waste building; the service building heating, ventilation, and air conditioning (HVAC) system; and the service air system were added as a result of the expert panel's review of recent NRC inspection reports. These systems were added to the scope on June 12, 1997, and a historical review of their performance was performed.

The team found the failure of the expert panel to include the six systems within the scope of the Maintenance Rule on July 10, 1996, to have been in violation of 10 CFR 50.65(b)(2). The team found that the licensee's actions upon identifying the omission of the systems were appropriate. Those actions included a review of the SSC performance histories, the review of preventive maintenance activities for those SSCs, the establishment of performance measures, and the determination that the omission was due to a lack of understanding of the Maintenance Rule when those SSCs were not included in the scope. The team also found that the omission of these systems did not have any effect with respect to SSC operability or categorization. The team noted that none of the added systems had exceeded any performance measures, either before or after being placed in scope. This licensee identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-458/9709-01).

c. Conclusions

Scoping of SSCs for inclusion in the Maintenance Rule program was generally acceptable. A noncited violation was identified for the failure to include six SSCs in the scope of the Maintenance Rule program on July 10, 1996.

M1.2 Safety or Risk Determination

a. Inspection Scope (62706)

The team reviewed the methods and calculations that the licensee's engineers had established for making the required safety determinations, including the probabilistic safety assessment (PSA) and associated modeling. The team also reviewed the safety determinations that were made for the functions that the team reviewed in detail during this inspection.

The team reviewed a sample of nonrisk-significant SSCs to assess if the safety significance was adequately established.

b. Observations and Findings

The team found that the expert panel had generally identified and classified SSCs as risk significant and nonrisk significant in accordance with Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, and NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2.

The team found that the expert panel had generally established performance measures that were supported by the assumptions of the PSA. This was accomplished by using the failure rates and unavailabilities assumed in the PSA to establish the performance measures, rather than establishing the performance measures first and then establishing the linkage of the measures to the PSA assumptions.

The team noted that the expert panel used a measure for reliability called the conditional probability [of success] for risk-significant, standby SSCs (i.e., high pressure core spray, reactor core isolation cooling, emergency diesel generators, residual heat removal, and fire protection). While the team understood the concept of a conditional probability [of success] as used by the licensee's PSA personnel, the team identified numerous concerns regarding the applicability to the Maintenance Rule as a measure of both reliability and availability.

Procedure PEP-0219, "Reliability Monitoring Program," Revision 5, was the governing procedure for the implementation of the Maintenance Rule at the River Bend Station.

To better understand the team's concerns, the following definitions from Procedure PEP-0219 are given.

> Availability - The time that an SSC is capable of performing its intended function as the total time that the intended function may be demanded. Availability is the numerical complement of Unavailability. See definition of Standby Availability for Availability of standby SSCs.
>
> Conditional Probability [of Success] - The probability a standby SSC will automatically start and run for a defined mission time, given a valid demand. This is the product of Standby Availability times the Probability of Starting times the Probability of Running.
>
> Reliability - A measure of the expectation that an SSC will perform its intended function upon demand, assuming the SSC is available, at any further instant in time; or, the probability a standby SSC will automatically start and run for a defined mission time, given a valid demand. This value is normally known as "Conditional Probability [of Success]."

In Guide EDG-PR-0001, "Reliability Monitoring Program," Revision 1, the guideline for implementing Procedure PEP-0219, reliability is equated to conditional probability [of success]. In paragraph 4.5.8, the team noted that in order to determine performance measures based on availability, the licensee personnel were to ". . . determine SSC Reliability (Conditional Probability [of Success]) . . . . " Also, in the note following paragraph 4.5.8, the team observed the equation for determining reliability (conditional probability [of success]) was presented as:

$$R = (A)(P_s)(P_r)$$

Where:

R = Target reliability for performance criteria consideration
A = Standby availability
P_s = Probability of start assuming one failure to start in 50 start demands
P_r = Probability of run
$P_r = e^{-\lambda t}$, where:
    λ = failure rate in failures per hour assuming one failure per year
    t = PSA mission time.

Additionally, Guide EDG-PR-0001, Paragraph 4.8.6 restates that "[r]eliability (or [c]onditional [p]robability [of success]) for the standby SSC is the product of the Standby Availability times the Probability of Start times the Probability of Run." Again, the team determined that reliability and conditional probability [of success] were synonymous.

On the basis of these definitions, the team questioned the licensee's representatives about the apparent lack of an availability performance measure for those five risk-significant, standby SSCs. The licensee representatives stated that conditional probability [of success] was a measure of availability because standby availability was one of the variables used to determine the conditional probability [of success].

Because the conditional probability [of success] was a combination of availability and reliability, the team found it difficult to understand how balancing of availability and reliability would be accomplished in accordance with 10 CFR 50.65(a)(3). Of particular concern was the potential masking of one factor by another. For example, reliability could increase as the result of vendor-provided improvements in design, and availability could decrease as the result of performing more preventive maintenance, thereby, offsetting the increase in reliability in the formula. If an SSC had all three factors at 90 percent, the conditional probability [of success] of the system would be 72.9 percent (i.e., 0.9*0.9*0.9=0.729). However, using the commonly adopted methodology, the performance measure for availability would be 90 percent and the performance measure for reliability would be 80 percent (i.e., 1-((1-0.9)+(1-0.9))=0.8) (assuming that the probability distribution is the same, i.e., exponential). If the availability decreased 5 percent, and the probability of run increased 5 percent, then the conditional probability [of success] would be 72.7 percent (i.e., 0.85*0.95*0.9=0.727), with the commonly adopted methodology values being 85 percent for availability and 85 percent (i.e., 1-((1-0.95)+(1-0.9))=0.85) for reliability. The commonly adopted methodology values, therefore, would result in a larger change in reliability, and possibly result in the identification of an imbalance between availability and reliability, whereas the change in the conditional probability [of success] would be negligible and could mask an imbalance between availability and reliability.

In the supplemental information (Attachment 2, Enclosure 1) provided by a licensee representative, the licensee explained how conditional probability [of success] was derived. The team reviewed this explanation, noting that the reasoning was the same as the team understood from the onsite inspection. The team requested the NRC PSA experts review the supplemental information and to provide feedback to the team. The NRC PSA personnel noted that the determination of the conditional probability [of success] was obtained by the multiplication of the probabilities of different probability distributions. That is, the probability for availability was a continuous distribution, while the probability to start was a binomial distribution. The team found, therefore, that the issue of conditional probability [of success] still warranted further review by the NRC to fully understand the concept of conditional probability [of success] and its applicability to the Maintenance Rule.

The team found, therefore, that the use of conditional probability [of success] had the potential to mask adverse trends in system performance because of the combination of the availability and reliability measures into one measure. The team identified this issue as an inspection followup item (50-458/9709-02) requiring further NRC review.

The team noted that the expert panel had established a reliability performance measure, supported by the PSA, of less than or equal to two maintenance preventable functional failures (MPFFs) per cycle for the fuel transfer system. The team questioned the expert panel about their deliberations for determining this measure. The team determined that a fuel handling accident had not been considered in the establishment of this measure. The expert panel acknowledged this oversight and revised the measure to be zero MPFFs that result in a fuel handling accident. The team found that this oversight was due to a lack of adequately identifying the functions for the fuel handling systems. The team also found that, while the performance measure for fuel handling was lacking, the significance was minimal because no fuel handling accidents had occurred and the level of monitoring had not changed.

c. Conclusions

The risk-significance ranking process was generally comprehensive. The licensee's use of the conditional probability [of success] had the potential to mask adverse trends in system performance because of the combination of the availability and reliability measures into one measure. This was identified as an inspection followup item. The omission of a fuel handling accident consideration in the development of performance measures for the fuel handling system was a weakness.

M1.3 Periodic Evaluation

a. Inspection Scope (62706)

The team reviewed the plans and procedures the licensee had established to ensure this evaluation would be completed every refueling cycle. The team also discussed these plans with the licensee's Maintenance Rule coordinator who was responsible for performing this evaluation.

b. Observations and Findings

The licensee, at the time of the inspection, had not been required to issue a periodic evaluation in accordance with 10 CFR 50.65(a)(3). The team found that guidance provided in Procedure PEP-0219 was adequate to meet the regulatory requirements.

M1.4 Balancing Reliability and Unavailability

a. Inspection Scope (62706)

The team reviewed the plans and procedures the licensee had established to ensure that the balancing of reliability and availability will be completed. The team discussed these issues with the licensee's representative who was responsible for performing these evaluations.

b. Observations and Findings

The team found that, for the risk-significant, normally operating SSCs, appropriate availability and reliability measures had been established. The team also found that the procedures for balancing availability and reliability for those SSCs met the intent of 10 CFR 50.65(a)(3).

However, as discussed in Section M1.2b., a conditional probability [of success] performance measure was established for five risk-significant, standby systems. As discussed below, the team found, and the licensee engineers agreed, that the performance measure for the residual heat removal system, 86 percent, was not conservative. Also, the performance measure for the high pressure core spray system was not supported by the PSA assumptions.

The nonconservative character for the residual heat removal system performance measure was the result of using a nonconservative assumption for the probability to run, 90 percent, in the calculation of the conditional probability [of success] limit. The team noted that the 90 percent value was based on an assumption of one failure in 3 years. The current operating experience for the residual heat removal system was a 100 percent probability to run for the last 9 years. Guide EDG-PR-0001, Paragraph 4.8.5.b, states, that "[i]f the number of run hours [were] too few, use a 'generic' failure rate for this type of component until sufficient data can be collected." In November of 1995 when the initial performance assessment of the residual heat removal system was done, 7 years of plant data showed a 100 percent probability to run.

The team noted that the conditional probability limit (performance measure) for the high pressure core spray system was 93.7 percent. However, the current conditional probability was 94.9 percent and the total probability of system success in the PSA was 97.1 percent. Again, the current data did not support the PSA in that the conditional probability limit and the current conditional probability were less than the PSA value. The team determined, from an historical review, that, in the initial 10 CFR 50.65 performance assessment of the high pressure core spray system, the licensee used the wrong value for the system availability (97.1 percent) instead of the actual availability (99 percent). The team found this error to be associated with the use of the conditional probability [of success]. This weakness will be addressed during the followup efforts for Inspection Followup Item 50-458/9709-02.

Procedure PEP-0219, Paragraph 6.4.6, requires specific performance measures for risk-significant SSCs be established to ensure the reliability and availability assumptions used in the plant-specific PSA are maintained or adjusted. In each of the above examples, the PSA assumptions were neither maintained nor adjusted. The licensee representatives agreed to correct these observations in its periodic evaluation to be done in November 1997 and review all SSCs for similar problems.

c. Conclusions

Reliability and availability performance measures for the risk-significant, normally operating SSCs were adequate to balance availability and reliability. The weaknesses that were identified were considered to be related to the licensee's use of the conditional probability [of success], which will be reviewed as part of Inspection Followup Item 50-458/9709-02.

M1.5 Plant Safety Assessments Before Taking Equipment Out-of-Service

a. Inspection Scope (62706)

The team reviewed the licensee's process for assessing the impact of removing equipment from service to support maintenance activities. The team reviewed the licensee's procedures and discussed the process with the Maintenance Rule coordinator, the expert panel members, operators, and maintenance schedulers. A sample of plant configuration changes that resulted from schedule changes and equipment failures was reviewed. The team then evaluated the licensee's assessment of the difference in risk as a result of the changes.

b. Observations and Findings

Although the equipment out-of-service (EOOS) computer program for online maintenance had been in place and relied upon to determine the online maintenance risk, the team noted that the senior scheduler (a former reactor operator) considered other factors, such as the use of the blended approach described in the EPRI PSA applications guide. The team observed that the blended approach consisted of combining the quantitative results from the EOOS computer program with engineering judgment and operating experience.

The team noted that the scheduler frequently ran calculations using the EOOS computer program as necessary to evaluate the amount of risk and that, if any questions arose, particularly for emergent work, he contacted the PSA engineer for advice. The team found that, while there was no requirement to perform risk evaluations for emergent work, the scheduler generally provided the results of the EOOS computer program evaluations for the emergent work configurations to others using a standardized e-mail form. The EOOS computer program included certain balance-of-plant equipment, although, predominantly, it included the risk-significant SSCs identified in the PSA.

In the control room, the shift technical advisor had the major responsibility for ensuring that the online risk profile was maintained according to the schedule and also for calculating the risk impact for emergent work. The shift technical advisor used the EOOS computer program, on the plant's local area network, to input equipment unavailability information.

No high risk combinations of equipment out-of-service were identified by the team as a result of the review of the control room logs for the period January through February 1997. The team found the personnel interviewed, particularly the senior scheduler and the shift technical advisor, to have a strong sense of involvement and ownership with the online maintenance program.

The team found the shutdown risk program was very comprehensive and conservatively used in conjunction with shutdown operation protection plans developed for each specific outage.

c. Conclusions

The licensee's online and shutdown maintenance risk-assessment programs were well developed and implemented, with advanced use of the EOOS computer program for both online and shutdown risk assessment. The persons interviewed expressed good knowledge of the risk-assessment process and a strong sense of involvement and ownership.

M1.6 Goal Setting, Monitoring, Performance Measures, and Preventive Maintenance

a. Inspection Scope (62706)

The team reviewed program documents and records in order to evaluate the process that had been established to set goals and monitoring requirements in accordance with 10 CFR 50.65(a)(1) and to verify that preventive maintenance was effective in accordance with 10 CFR 50.65(a)(2) of the Maintenance Rule. The team also discussed the program with the Maintenance Rule coordinator, system engineers, plant operators, and schedulers.

The team reviewed the systems and components listed below to verify: that goals or performance measures were established with safety taken into consideration; that industry-wide operating experience was considered for goal setting, where practical; that appropriate monitoring and trending were performed; and that corrective action was taken when an SSC function failed to meet its goal or performance measure, or experienced an MPFF. The team also reviewed condition reports and equipment history for these systems from July 1996 to the beginning of this inspection.

SYSTEM DESIGNATOR	SYSTEM NAME
PCI	Primary containment isolation
051	Nuclear boiler instrumentation
107	Reactor feedwater
115	Reactor plant component cooling water
130	Normal service water
202/109	Automatic depressurization system/ safety relief valves
203	High pressure core spray
204	Residual heat removal
209	Reactor core isolation cooling
256	Standby service water
303	480 Vac electrical distribution
304	120 Vac electrical distribution
305	125 Vdc distribution
402/410	Control building HVAC (HVC) and HVAC chillers (HVK)
552	Containment atmosphere and leakage monitoring
N/A	Structures

b. Observations and Findings

The team noted that the PCI, reactor feedwater, reactor plant component cooling water, high pressure core spray, HVC/HVK, and 120 Vac distribution systems' performances were such that the SSCs were being monitored in accordance with 10 CFR 50.65(a)(1). The team found, generally, that appropriate corrective actions had been taken to address the causes of the unacceptable performance and that appropriate goals and monitoring measures had been established for each of these systems.

PCI

The team found the performance measures for the PCI system included a leakage limit of $<0.6L_a$ for any Type B or C valve, as defined in Appendix J to 10 CFR Part 50 (where L_a is the design allowable leakage rate for containment operability to meet 10 CFR Part 100 limits). The regulatory limit, as defined in Appendix J and the River Bend technical specifications for all Type B and C valves, is $<0.6L_a$. Before the Maintenance Rule was issued, the licensee established a testing program to meet the requirements of Appendix J. If the performance of a valve decreased to an Appendix J administrative limit, then the Appendix J program would cause the licensee's staff to take corrective action. The administrative limits employed by the licensee's staff for the Appendix J testing program varied from valve to valve (e.g., from 20 sccm/min for Valve KJB-Z20 to 2,000 sccm/min for Valve CCP-MOV138).

In order for a valve to exceed the performance measure established for the Maintenance Rule program, its leakage would have to exceed 66,420 sccm/min. As a result, the use of either $0.6L_a$ for any one valve or for total leakage would not demonstrate that the performance or condition of the valves was being effectively controlled through the performance of appropriate preventive maintenance until a regulatory limit had been reached or exceeded. The team determined this to be the first example of a violation of 10 CFR 50.65(a)(2) in that the ability of preventive maintenance to assure the reliability of the containment isolation valves was not demonstrated (50-458/9709-03).

In the supplemental information (Attachment 2, Enclosure 4) provided after the onsite inspection, the licensee's representatives restated the position that was presented during the inspection. Specifically, that the performance measure for leakage being $\leq 0.6L_a$ for any one Type B or C valve was acceptable because it "provides a conservative indication of the effectiveness of maintenance before the function is lost." While it was stated in the supplemental information that the expert panel was considering changing the performance measures to measures that would identify degrading maintenance performance prior to a regulatory limit being reached, the team found that the expert panel did not acknowledge that the establishment of performance measures at a regulatory limit did not meet the intent of the Maintenance Rule.

As stated above, the team considered the performance criteria being set at $\leq 0.6L_a$ for any one Type B or C valve was inappropriate to establish that the performance or condition of the PCI system was being adequately controlled by the performance of appropriate preventive maintenance. This position was based on the fact that, using the measure established by the licensee, all Type B and C valves could experience leakage up to $0.6L_a$ without the Maintenance Rule program being implemented. The team found, therefore, that the performance measure, as documented at the time of the inspection, could result in the loss of function (e.g., containment integrity) before the measure would be exceeded.

During review of the testing history of the PCI system, the team found previously unidentified MPFFs. The team found five valve leakage failures in which floor drain valves' leakage had exceeded $0.6L_a$ because of line debris. It was the team's finding that these were MPFFs in that cleaning of the lines could have been performed as preventive maintenance. The five valves were identified on maintenance work order requests, with one exception. The maintenance work order requests were 203824, 205995, 303692, and 304304 with Condition Report 96-0296. The team found the failure to perform adequate cause determinations for these valves to be an example of an inadequate monitoring program, which failed to demonstrate that the performance or condition of SSCs was being effectively controlled through appropriate preventive maintenance such that the SSCs remained capable of performing their intended functions. This was the first instance of the second example of a violation of 10 CFR 50.65(a)(2) (50-458/9709-03). *This example was indicative of the lack of understanding* and clear guidance on determining functional failures and MPFFs.

In the supplemental information (Attachment 2, Enclosure 6), the licensee's representatives acknowledged that the initial classification of the five check valve failures as only functional failures was incorrect, and that the failures should have been classified as MPFFs. The licensee's representatives wanted to take credit for actions taken in the 1995-1997 time frame. However, while in accordance with the corrective action program in effect at the time of the failures, those actions did not include identifying the root cause(s). The failure to identify the failures as MPFFs resulted in the licensee's engineers not identifying the root cause(s) (e.g., the source of the trash in the drain lines), taking corrective actions commensurate with safety (e.g., establishing a preventive maintenance task to clean the drain lines), establishing goals, and establishing monitoring requirements in accordance with 10 CFR 50.65(a)(1). Without the root cause(s) being identified, the establishment of adequate goals could not be assured.

Nuclear Boiler Instrumentation

The inspectors noted that, at the time of the inspection, there was no performance measure for availability for the nuclear boiler instrumentation (primary plant instrumentation, e.g., reactor vessel level, drywell pressure, recirculation flow, etc.) even though the system was designated risk-significant. The team was informed that, even though the system remained in Category (a)(2), the risk-significance was changed on September 4, 1997, and that the determination of appropriate performance measures had not been completed. As such, the licensee engineers could not demonstrate that the performance or condition of the nuclear boiler instrumentation system was being effectively controlled through the performance of appropriate preventive maintenance.

In accordance with 10 CFR 50.65, an SSC may be monitored in accordance with 10 CFR 50.65(a)(2) if its performance or condition has been demonstrated to have been effectively controlled through the performance of appropriate preventive maintenance. If such a demonstration has not been made, then the SSC shall be assessed for monitoring in accordance with 10 CFR 50.65(a)(1).

While the changing of risk-significance was not specifically addressed in Procedure PEP-0219, the team considered the change to be equivalent to not meeting a performance measure because there were no performance measures established in accordance with Procedure PEP-0219 for the high risk-significant nuclear boiler instrumentation system, in particular, there was no measure for availability.

Procedure PEP-0219 requires that a cause determination of appropriate depth be performed. This evaluation is to be done via the condition reporting program, which has timeliness goals established, commensurate with safety (risk-significance).

On September 4, 1997, when the expert panel changed the risk classification of the nuclear boiler instrumentation system, the system engineer was directed to establish new performance measures. No condition report was initiated from that time to the end of the onsite inspection to address the need for new performance measures, the need to review past performance history, and to demonstrate that the performance or condition was being adequately maintained by the performance of appropriate preventive maintenance.

The team noted that, as of October 31, 1997, appropriate performance measures had not been developed in a timely manner for a high-risk-significant system to demonstrate that the performance or condition of that system was being effectively controlled through the performance of appropriate preventive maintenance.

Because the nuclear boiler instrumentation system's performance or condition was neither being monitored in Category (a)(2) with appropriate performance measures to demonstrate effective control through appropriate preventive maintenance nor being monitored in Category (a)(1), the team identified this as the first instance of the third example of a violation of 10 CFR 50.65(a)(2) (50-458/9709-03).

Reactor Feedwater System

The team identified a loss-of-functional failure that occurred to the reactor feedwater system; however, the system engineer and the expert panel did not agree with the team's determination. In Condition Report 96-1877, a licensee's employee reported a failure of the copper tubing actuating instrument air line to a feed pump minimum flow recirculation valve. The tubing failure caused the valve to fail open and resulted in a slight reactor power reduction (2-4 percent) due to reduced feed pump output. This power reduction was necessary to maintain reactor vessel water level constant with the diversion of flow through the minimum flow recirculation valve. This reduction in reactor vessel water level resulted in the reactor feedwater system not meeting its Maintenance Rule function to maintain level to support 100 percent power operation.

The team noted that the first function listed for the feed system was to provide sufficient feedwater flow to maintain the required reactor vessel level for all conditions. Licensee representatives were not able to explain why an inability to maintain reactor vessel level at full power was not identified as a functional failure or a degraded function. The team found that the failure to identify this as a functional failure had no effect, from a Maintenance Rule perspective, because the failure was not maintenance preventable. However, the team found this failure to identify a functional failure to be an example of the licensee's difficulty in identifying loss of function or degraded functions. Aside from this one example, the team determined that the feedwater system was appropriately monitored by the licensee's Maintenance Rule program.

Reactor Plant Component Cooling Water System

The team noted that licensee personnel had identified that reactor plant Component Cooling Water Pumps CCP-P1A and CCP-P1C experienced high unavailability due to repetitive pump overhauls, which were necessitated by high bearing vibrations. The system engineer evaluated the times of unavailability and determined that, at no time, did the system have less than two operating pumps. However, licensee personnel recognized that, while the system was still meeting the established performance measures, the pump unavailability information was not being properly assessed. The team found the monitoring of the reactor plant component cooling water system at the system level was due to a lack of understanding the intent of the Maintenance Rule.

In order to avoid the masking effect where a good performing pump might mask the poor performance of another pump, the expert panel, during the second quarter of 1997, established an 18-month sliding average pump availability goal of ≥99.2 percent for each pump. Since two of the pumps had experienced high unavailability rates, and they did not meet the new pump performance measures, plant engineering initiated Condition Report 97-1198 on August 12, 1997, to evaluate the system for Category (a)(1) status.

The evaluation concluded that Pump CCP-P1C had been continuously available for 18 months; however, Pump CCP-P1A was not meeting its availability requirement and was, therefore, evaluated for the establishment of goals and monitoring requirements in accordance with 10 CFR 50.65(a)(1). This information, along with the corrective actions taken to prevent recurrence, was presented to the expert panel on September 4, 1997. The expert panel determined that the performance had not met the performance measures, that the cause had been determined, and that actions had been taken to correct the cause and prevent recurrence. The team found that the expert panel had appropriately evaluated the performance and had established appropriate goals and monitoring measures for the reactor plant component cooling water system.

Service Water System

Similar to the reactor plant component cooling water system, licensee personnel recognized that, while the service water systems were meeting the established performance measures, a potential existed for not recognizing pump unavailability information due to a masking effect. In order to avoid a masking effect, the expert panel, during the second quarter of 1997, established a 18-month sliding average pump availability goal of ≥99.2 percent for each pump in the normal service water system, and ≥99.4 percent for each pump in the service water cooling system. The actual performance was then compared to the new performance measures. The team noted that the actual performance met the measures.

120 Vac Electrical Distribution System

The licensee's 120 Vac electrical distribution system was a risk-significant system and included eight uninterruptible power supplies. Two of the units were safety-related, while six were not. Six of the uninterruptible power supplies (including the two safety-related units) were included within the scope of the Maintenance Rule program. One of the nonsafety-related uninterruptible power supplies (BYS-INV-06) had experienced repetitive MPFFs due to failed fuses on August 14, 1994, October 25, 1995, and July 24, 1996. As a result of these failures, the system was appropriately dispositioned in accordance with 10 CFR 50.65(a)(1).

At the time of the inspection, the licensee had determined those corrective actions to address the fuse failures were effective, based on a lack of recurrence. Nonetheless, the licensee determined that the monitoring established in accordance with 10 CFR 50.65(a)(1) was still necessary due to additional, unrelated functional failures; the static switch had auto-transferred to bypass on June 13 and July 15, 1997 (the static switch auto bypasses the power source around the inverter). The licensee had not determined the cause of the failures at the time of the inspection. The team found the continued monitoring due to the unexplained failures to be appropriate.

Although monitoring of Inverter BYS-INV-06 was appropriate, establishment of goals and monitoring requirements for the system were not completed until November 16, 1996, almost 4 months after the effective date for implementation of the Maintenance Rule. The licensee's representative stated that establishment of goals and monitoring measures were delayed due to troubleshooting, repair, and cause determination.

As with the nuclear boiler instrumentation system discussed above, the team noted that the risk-significance of the 120 Vac electrical distribution system had been changed on September 4, 1997, and that the determination of appropriate performance measures had not been completed. As such, the licensee engineers could not demonstrate that the performance or condition of the 120 Vac electrical distribution system were being effectively controlled through the performance of appropriate preventive maintenance.

In accordance with 10 CFR 50.65, an SSC may be monitored per Category (a)(2) if its performance or condition has been demonstrated to have been effectively controlled through the performance of appropriate preventive maintenance. If such a demonstration has not been made, then the SSC shall be assessed for monitoring in accordance with 10 CFR 50.65(a)(1).

While the changing of risk-significance was not specifically addressed in Procedure PEP-0219, the team considered the change to be equivalent to not meeting a performance measure because there were no performance measures established in accordance with Procedure PEP-0219 for a high-risk-significant 120 Vac electrical distribution system, in particular, there was no measure for availability.

Procedure PEP-0219 requires that a cause determination of appropriate depth be performed. This evaluation is to be done via the condition reporting program, which has timeliness goals established, commensurate with safety (risk-significance).

On September 4, 1997, when the expert panel changed the risk classification of the 120 Vac electrical distribution system, the system engineer was directed to establish new performance measures. No condition report was initiated from that time to the end of the onsite inspection to address the need for new performance measures, the need to review past performance history, and to demonstrate that the performance or condition was being adequately maintained by the performance of appropriate preventive maintenance.

The team noted that, as of October 31, 1997, appropriate performance measures had not been developed in a timely manner for a high-risk-significant system to demonstrate that the performance or condition of that system was being effectively controlled through the performance of appropriate preventive maintenance.

Because the 120 Vac electrical distribution system's performance or condition was neither being monitored in Category (a)(2) with appropriate performance measures to demonstrate effective control through appropriate preventive maintenance nor being monitored in Category (a)(1), the team identified this as the second instance of the third example of a violation of 10 CFR 50.65(a)(2) (50-458/9709-03).

HVC and HVK Systems

During the expert panel meeting on August 21, 1997, the panel reviewed the status of HVC and HVK systems. The expert panel approved the addition of a new goal, which stated that at least one chiller on each division must pass two consecutive surveillance tests for heat removal capacity (greater than design requirements) with a test frequency greater than 6 months. The expert panel also approved analyzing the combined HVC and HVK system as separate entities such that monitoring of the HVC system could be stopped, while continuing to monitor the HVK system. The reasoning for this new approach was because the system problems were related to the water side of the system (i.e., low heat removal capacity of the chillers). Three of the four chillers did not successfully pass the surveillance test procedure for heat removal capacity. Corrective actions were being implemented, including chemical cleaning of each chiller.

Containment Atmosphere and Leakage Monitoring System

The containment atmosphere and leakage monitoring system utilized a variety of instrument trains to monitor containment parameters. Those parameters were hydrogen concentration, radiation level, pressure, differential pressure, and temperature. Additionally, the containment atmosphere and leakage monitoring system monitored suppression pool temperature and level. Monitoring against established performance measures was performed at the system level, except for the hydrogen analyzers, which were monitored at the train level. At the time of the inspection, the expert panel considered all portions of the system to be reliable. The containment atmosphere and leakage monitoring system was deemed to be low-risk significant by the expert panel.

Between March 17, 1995, and October 29, 1997, the hydrogen analyzers experienced a total of nine surveillance failures due to excessive instrument drift (eight of the failures occurred in the last 13 months). Six of the failures were attributable to Division II while three failures were associated with Division I. In all cases, instrument accuracy had drifted beyond acceptable limits during the as-found calibration check, which utilized a 10 percent concentration sample standard. While, in five of the cases, the drift was slightly outside the acceptance band, in other cases the deviation was much more significant (indicating as low as 5.5 percent hydrogen concentration when measuring the 10 percent sample standard).

In July 1997, the licensee identified that the hydrogen analyzers were not being monitored against appropriate performance measures. Specifically, the analyzers were being monitored with the remainder of the containment atmosphere and leakage monitoring system, and at the system level. Since the analyzers were actually standby instruments, monitoring should have been performed at the train level. The expectation that the monitoring of low-risk, standby systems should be at the train level is given in Regulatory Guide 1.160, which endorses NUMARC 93-01.

In response to the above finding, the licensee established performance measures for each train (95 percent availability) and immediately determined that the trains did not meet the performance measures. The Division I train availability was 92.5 percent and Division II train availability was 93.6 percent. At that time (July 1997), seven failures were on record.

Upon exceeding the established performance measures, the containment atmosphere and leakage monitoring system engineer initiated a cause determination to address the failure to meet the noted performance measures. The documented cause determination concluded that the performance problems had been resolved and no additional monitoring was necessary (although the root cause of the instrument drift failures was not specifically addressed). Corrective measures cited in the document included replacing several parts (in both units) and increasing the surveillance interval to monthly for the Division II unit (the most problematic analyzer). The cause determination also noted that there were no additional failures associated with either unit for almost 3 months.

During subsequent discussions with the NRC inspectors, the containment atmosphere and leakage monitoring system engineer was unable to identify the components that were replaced during maintenance or how the repairs corrected the problems. Additionally, the system engineer stated that he believed the failures to be caused by a design inadequacy.

Of particular interest, the cause determination stated that all the known failures were Maintenance Rule functional failures, but none were considered MPFFs. The document went on to state that all of the maintenance specified by the maintenance program was performed. As such, neither the system engineer nor the expert panel considered any of those failures to have been maintenance preventable.

On September 4, 1997, the expert panel considered establishing goals and monitoring requirements for the hydrogen analyzers and discussed the problems with the system engineer. During the meeting, the expert panel did not discuss the nature and repetitiveness of the failures. The conclusion of the expert panel was that system performance was acceptable and no additional monitoring was warranted.

On September 11, 1997, the Division I hydrogen analyzer experienced another failure due to instrument drift. On October 29, 1997, the Division II unit experienced a similar failure.

Although the licensee had initially identified that the hydrogen analyzers were not being monitored at the train level prior to July 1997, corrective measures since that initial finding were not effective and additional failures occurred. Additionally, the inspectors identified that the licensee had failed to demonstrate acceptable performance of the hydrogen monitors through the performance of appropriate preventive maintenance. This issue was identified as the second instance of the second example of a violation of 10 CFR 50.65(a)(2) (50-458/9709-03).

In the supplemental information provided (Attachment 2, Enclosure 2), the licensee representatives provided the same information as discussed above. The team noted that the licensee representatives considered that "the intent and the spirit of the Maintenance Rule was and is being maintained." The team found this statement to be incorrect because the actions initially taken, after the performance was identified as inappropriate, did not identify the need to identify the root cause(s), take action to correct and prevent recurrence, and establish goals against which the performance would be measured to evaluate the effectiveness of those corrective actions. While the calibration failures were identified on condition reports, and some components were replaced, no tie was established between the replaced components and the failures that were observed. The team found that the root cause of the failures was not identified and that effective corrective actions were not implemented.

The team considered the following to be contributors to the violation:

- Failure to recognize (early in the implementation of the program) that performance monitoring should be performed at the train level for the hydrogen monitors. Train level monitoring was not initiated until July 1997.
- An inadequate cause determination, in that the document failed to appropriately identify multiple and repetitive MPFFs. The licensee erroneously assumed that completion of all scheduled maintenance precluded failures from being considered maintenance preventable.
- Failure on the part of the expert panel to thoroughly discuss the nature of problems associated with the hydrogen monitors when the goals and monitoring requirements were considered on September 4, 1997. A more thorough discussion would have likely revealed that repetitive MPFFs had occurred and that goals and monitoring were required in accordance with 10 CFR 50.65(a)(1).

Structures

Licensee personnel had documented a Maintenance Rule inspection and evaluation of structures in Engineering Report M/C-96-002, issued on April 17, 1996. The report documented findings of the inspections and most findings were dispositioned use-as-is. However, the report did not provide a basis for the disposition of the findings. Also, the report indicated that the attributes identified in three existing surveillance procedures were used to perform the inspections. However, the team could not determine which attributes were used to perform the baseline inspection for each structure because inspection records and checklists were not available. There were two condition reports generated as a result of the inspections. One report identified rusted steel, mostly fasteners and washers, in the drywell. The other identified concrete cracks in the reactor building.

The team noted that the turbine building was inspected and seven minor deficiencies were identified on January 10, 1996. However, the turbine building was not placed in the scope of the Maintenance Rule until June 1997. Conversely, the radioactive waste building was placed in the scope of the rule in June 1997, but it had not been inspected at that time. The team also noted that all baseline inspections of structures in the scope of the Maintenance Rule on July 10, 1996, and the turbine building, were performed and completed during the period January 10-14, 1996. (Section M1.1 of this report discusses the enforcement disposition related to the program scoping of the turbine and radioactive waste buildings.)

The licensee's program for monitoring the effectiveness of maintenance on structures was implemented by Procedure PEP-0219 and conducted in accordance with Guide EDG-PR-0001. According to the process described in these documents, structures were not assessed for risk significance. The team noted that in mid-1997, the drywell, containment, and standby cooling tower were identified as risk significant by a subgroup of the expert panel using a subjective ranking process.

The team determined that a structural functional failure was defined as an unacceptable condition as determined by the Civil Structural Group in accordance with Section 6.8.3.2 of Procedure PEP-0219. The team was told by licensee representatives that the structural functions for those structures in the Maintenance Rule program were to provide environmental protection or structural support for SSCs within the structures that were in the Maintenance Rule program. While Procedure PEP-0219 did not specify what constituted an unacceptable condition, the licensee representatives stated that a condition would be considered unacceptable when the structures were not capable of providing environmental protection or structural support for the Maintenance Rule program SSCs within the structure. Without any other definition of unacceptable condition provided, the team considered that the licensee's position was that a functional failure would be identified only after catastrophic failure (e.g., a "failure to meet all design basis loading requirements" [Attachment 2, Enclosure 5], or a release of radioactive material in excess of 10 CFR Part 100 limits) of the structure.

The team found, therefore, that there were no provisions for setting of goals until catastrophic structural failure occurred. This method of monitoring and evaluating did not assure that the functions would be maintained by preventive or periodic maintenance and surveillance and was identified as the fourth example of a violation of 10 CFR Part 50.65(a)(2) (50-458/9709-03). The team also found this example to be indicative of a lack of understanding of the intent of the Maintenance Rule, in that, a functional failure was not possible, in accordance with the licensee's program, until a catastrophic failure of a structure occurred.

In the supplemental information (Attachment 2, Enclosure 5) provided after the onsite inspection, licensee representatives provided information to explain why the program for structural monitoring was acceptable. The team did not identify any new information that indicated how the structural monitoring program would identify degrading performance or condition such that the structure would be assessed for the establishment of goals and monitoring requirements prior to catastrophic failure.

c. Conclusions

Four examples of a violation of 10 CFR 50.65(a)(2) were identified for the failure to demonstrate that the performance of SSCs was effectively controlled by the licensee's preventive maintenance efforts. The licensee's monitoring of the effectiveness of maintenance on structures within the scope of the licensee's Maintenance Rule program lacked necessary detail and did not provide an adequate process to transition from (a)(2) to (a)(1). These examples were indicative of a lack of understanding of the intent of the Maintenance Rule by the licensee's staff.

M2 Maintenance and Material Condition of Facilities and Equipment

a. Inspection Scope (62706)

In the course of verifying the implementation of the Maintenance Rule using Inspection Procedure 62706, the team performed in-plant walkdowns to examine the material condition of the following systems and components: HVC/HVK, reactor core isolation cooling, reactor feedwater, residual heat removal, reactor plant component cooling water, and 480 Vac distribution.

b. Observations and Findings

No external corrosion or leaks were identified during any of the field inspections. Minor housekeeping issues were identified in the Train B residual heat removal room. Those issues were promptly corrected by the licensee's representatives accompanying the team member.

c. Conclusions

No significant material deficiencies were identified during the system walkdowns. Housekeeping was very good in the areas inspected, considering that the plant had been recently returned to operation after a refueling outage.

M3 Maintenance Procedures and Documentation

a. Inspection Scope

The team reviewed the Maintenance Rule Data Information System (MRDIS) that was maintained by the Maintenance Rule coordinator, for its contents.

b. Observations and Findings

The licensee's program in Guide EDG-PR-0001, Paragraph 2.13, requires that all Maintenance Rule data, such as systems in scope, functions, performance measures, and failure analysis, be documented in the MRDIS. The team noted problems with the definition of some system boundaries in the MRDIS. For example, the primary containment system, which included the secondary containment, did not have a boundary established between it and the nuclear steam supply shutoff system, which supplied the initiation signals and logic for the PCI valves. A licensee's representative took steps to revise the MRDIS to correct this interface problem with the primary containment system and nuclear steam supply shutoff system.

A system engineer stated that the 125 Vdc system included the 480 Vac supply breakers for the invertors and chargers. However, the description of the six subsystems in the initial 10 CFR 50.65 performance assessment for the 125 Vdc electrical distribution system dated October 25, 1995, and the addendum dated May 20, 1996, did not identify the supply breakers. The system engineer also stated that he did not know of an electrical one-line drawing of the 125 Vdc system.

The team found that the lack of clear boundary definitions was a weakness in the Maintenance Rule program.

The team observed that, in some cases, there was no clear, quantitative definition of what was a functional failure. Examples included:

- The PSA system notebook listed a differential pressure of 399 psid (2571 kPad), not 415 psid (2861 kPad). The licensee's representatives agreed with the team's observation and took actions to correct PSA system notebook. In addition, the team found that the most accurate quantitative definition of the design flow rate was found in a calculation that was not referenced in the MRDIS.

- The MRDIS functions did not identify the 37-second start requirement for the residual heat removal system that was found in the safety analysis report; nor did it include the 4-hour mission time for the 125 Vdc electrical distribution system found in the safety analysis report.

- All SSC functions had not been included in the MRDIS. The team noted that Emergency Operating Procedures, Enclosure 6, included a head spray function for the residual heat removal system that was not identified in the MRDIS. Licensee staff took actions to correct this omission and acknowledged the need to review all the functional data in the MRDIS.

- Some functions included the scope of the Maintenance Rule had no performance measures established. The team found the PCI system had two performance measures, one for isolation valves and one for air locks. The expert panel had not established performance measures for the functions to provide structural support of safety-related SSCs and to provide environmental protection of safety related equipment.

c. <u>Conclusions</u>

The governing procedure and guidelines for implementing the Maintenance Rule program were weak, as evidenced by the lack of clear boundary definitions, function definitions, and quantitative performance measures that contributed to the violation discussed in Section M1.6.

III. Engineering

E4 Engineering Staff Knowledge and Performance

a. Inspection Scope (62706)

The team interviewed the expert panel and engineering personnel to assess their understanding of the Maintenance Rule and associated responsibilities. The team also reviewed the training that had been administered to system engineering personnel.

b. Observations and Findings

The team found the panel to consist of experienced supervisory-level staff who had met consistently on a frequency ranging from once per week to once per month. The panel members indicated that both the system engineers and the expert panel members had been trained in both PSA and the Maintenance Rule. An expert in PSA was a member of the panel.

All system engineers interviewed were knowledgeable of their systems. Although not extensively trained in the PSA, the system engineers were knowledgeable of the necessary PSA information.

The team noted some differences among the system engineers as to what the system engineers' responsibilities were with respect to the Maintenance Rule. Some system engineers stated that they were responsible for the identification of functional failures only. Other system engineers stated that they were responsible for identifying MPFFs also.

c. Conclusions

The level of experience of the expert panel members and the frequency of the expert panel meetings were strengths in the implementation of the Maintenance Rule program.

The system engineers' knowledge was generally consistent with their Maintenance Rule program responsibilities.

V. Management Meetings

X1 Exit Meeting Summary

The team discussed the progress of the inspection on a daily basis and presented the inspection results to members of licensee management at the conclusion of the onsite inspection on October 31, 1997. In addition, a supplemental telephonic exit was held on January 20, 1998, to discuss the results of the in-office inspection of the supplemental information, and the enforcement findings from the inspection. The licensee's representatives acknowledged the findings presented and stated that they would review the issues to determine if they had any differing positions.

The team asked the licensee's staff and management whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

L. Ballard, Quality Supervisor
K. Barnes, Technical Specialist
R. Barnes, Supervisor, Reliability Programs
L. Bedell, Senior Engineer
O. Bulich, Rotational Manager, Plant Engineering Programs
P. Campbell, Technical Assistant, Nuclear Safety and Regulatory Assurance
R. Cole, Supervisor, Plant Engineering
D. Dormandy, Manager, Plant Engineering
J. Fowler, Manager, Quality Assurance
D. Gilley, Engineering Supervisor
R. King, Director, Nuclear Safety and Regulatory Affairs
J. Malara, Engineering Supervisor
R. McAdams, Nuclear Safety and Regulatory Affairs
J. Mead, Engineering Supervisor
A. Nguyen, Maintenance Rule Coordinator
W. O'Mally, Operations Manager
J. O'Neil, Senior Technical Specialist
M. Padgett, Administrative Assistant
A. Shahkarami, Manager, Engineering
P. Sicard, Manager, Safety and Engineering Analysis
M. Stevens, Mechanical Superintendent
J. Ward, Director, Nuclear Training
T. Watkins, Rotational Manager, Plant Engineering Systems

NRC Personnel

D. Powers, Chief, Maintenance Branch
W. Smith, Senior Resident Inspector

INSPECTION PROCEDURE USED

IP 62706: Maintenance Rule

ITEMS OPENED AND CLOSED

Opened

458/9709-01	NCV	Failure to include six SSCs within the scope of the Maintenance Rule program

458/9709-02	IFI	Applicability of the use of conditional probability [of success] in the Maintenance Rule program
458/9709-03	NOV	Failure to demonstrate adequate performance measures in accordance with 10 CFR 50.65(a)(2)

Closed

458/9709-01	NCV	Failure to include six SSCs within the scope of the Maintenance Rule program

PARTIAL LIST OF DOCUMENTS REVIEWED

CONDITION REPORTS:

95-0558	96-1229	96-2007	97-0944	97-1368
96-0346	96-1374	96-2081	97-0994	97-1395
96-0347	96-1442	96-2104	97-1039	97-1469
96-0512	96-1706	96-2116	97-1133	97-1620
96-0545	96-1877	97-0099	97-1172	97-1679
96-0631	96-1957	97-0424	97-1198	97-1909
96-0880	96-1983	97-0483	97-1228	97-1937
96-0883	96-1988	97-0489	97-1305	97-1946
96-0981				

PROCEDURES:

PEP-0219	"Reliability Monitoring Program," Revision 5
STP 704-3307	"Structural Settlement Monitoring," Revision 4
STP 057-3700	"Containment Structure Integrity Verification," Revision 5
STP 057-3701	"Drywell Structure Integrity Verification," Revision 5
STP 303-1601	"120 and 480 VAC Breaker Overload Functional Test," Revision 14
STP 254-4203/4204	"Drywell and Containment Hydrogen Analyzer Channel Calibration"

<u>MISCELLANEOUS</u>:

Engineering Report M/C-96-002, April 17, 1996

Engineering Report E/S-96-001, "Scope Determination and Initial Assessment for Maintenance Rule Implementation," Revision 0

Engineering Report R-SEA-97-001-00, "Shutdown Equipment Out of Service (SHEOOS) and Shutdown Probabilistic Safety Assessment (SHPSA)," April 30, 1997

Engineering Report R-SEA-97-007-00, "Probabilistic Safety Assessment and Maintenance Rule Performance Criteria," Revision 0

Engineering Calculation G13.18.12.3*171-1, "Shutdown Safety Function Defense in Depth Color Codes," Revision 1

Engineering Guide, EDG-PR-0001, "Reliability Monitoring Program," Revision 01

Individual Plant Examination - River Bend Station, Volumes 1 and 2, February 1, 1993

J.S. Miller, et al., "River Bend Nuclear Station Time Dependent Core Damage Frequency," Third Workshop on Living PSA Application, Hamburg, Germany, May 11-12, 1992

R. Christie and W. McDougald, "Impact of Safety Requirements on Component Availability," American Nuclear Society Conference, Boston, Mass., June 11, 1992

S. Holbert and R. Christie, "System Reliability Program at River Bend Nuclear Station," 1992 ANS/ASME Nuclear Energy Conference, San Diego, California, August 23-26, 1992

R. T. Kelly, et al., "Trending Emergency Diesel Generator Conditional Probability - River Bend Station - Gulf States Utilities," ASME-IEEE Joint Power Conference, Atlanta, Georgia, October 21, 1992

EPRI Technical Bulletin 96-11-01, "Monitoring Reliability for the Maintenance Rule," November 1996

EPRI Technical Bulletin 97-3-01, "Monitoring Reliability for the Maintenance Rule - Failures to Run," March 1997

Entergy memorandum to R.E. Barnes from P.A. Sicard, "Core Damage Frequency Graphs: Second Quarter - 1997," September 11, 1997

"On-Line Maintenance Guidelines," Revision 0

EPRI TR-105396, "PSA Applications Guide," August 1995

Control Room logs for the period January to February 1997

SYSTEM DESCRIPTIONS:

LOTM 3	Nuclear Boiler Process Instrumentation System
LOTM 46	Containment Atmosphere Monitoring
LOTM 56	AC Distribution

ATTACHMENT 2

SUPPLEMENTAL INFORMATION SUBMITTALS[1]

Enclosure 1	Conditional Probability (November 18, 1997)
Enclosure 2	Containment Atmosphere and Hydrogen Monitoring System (552) (November 18, 1997)
Enclosure 3	Gaitronics Power Supply (November 18, 1997)
Enclosure 4	Performance Criteria for the Primary Containment Integrity System (November 18, 1997)
Enclosure 5	Performance Criteria for Structures (November 18, 1997)
Enclosure 6	Containment Isolation Valves (CIV) Cause Determination (November 20, 1997)

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[1]Documents received via Internet from R. McAdams, Entergy, to C. Paulk, NRC Region IV. Content of documents not altered.

CONDITIONAL PROBABILITY

Introduction

This white paper clarifies the use and impact of conditional probabilities as performance criteria in the RBS Maintenance Rule program. RBS has used conditional probabilities as a means to balance availability and reliability of the HPCS, RCIC, RHR, and Emergency Diesel Generator systems. During the RBS Baseline Maintenance Rule Inspection from October 27 through 31, 1997, questions were raised about how specific values used to calculate conditional probabilities compare to PSA data. This white paper addresses these concerns and uses CDF estimates to show that the effects on CDF of the conditional probability performance criteria is within the guidance of the NEI PSA Applications Guide.

General Discussion

RBS uses conditional probabilities as a part of our Maintenance Rule Performance criteria for selected systems trended under our program. In this method, availability, start probability, and run probability are multiplied together to establish the conditional probability. For example the RCIC system conditional probability is 0.832. This is the product of the standby availability of 0.965, the probability of start of 0.980, and the probability of run of 0.880. Thus, the result is as follows:

$$0.965*0.980*0.880 = 0.832$$

Note that each of these numbers represent goals established at the system or train level. In general, they do not directly correlate to individual component failures in the PSA. As an example, the start probability above represents the start probability of the RCIC system. It is not intended to explicitly represent the start probability of the RCIC turbine (a component of the system) and thus would not directly compare to the failure to start probability of the RCIC turbine in the PSA. One can compare the maintenance unavailability from the PSA with that in the Maintenance Rule conditional probability method. In addition, the **probability of start** multiplied by the **probability of run** is a means of expressing the equipment reliability of the system in question. This also can be compared between the PSA assumptions and the Maintenance Rule conditional probability method.

The conditional probability method used at RBS provides balance between reliability and availability and also recognizes industry insights to establish the performance criteria. This method consists of the use of RBS performance indicator goals, based on the INPO SSPI program, expert panel judgment in approving the conditional probability performance criteria for each applicable system (RHR, HPCS, RCIC, and EDGs), and PSA insights in benchmarking the resulting conditional probability against the availability derived from the PSA. Taken together, these elements balance current industry practices, plant-specific expert judgement, and the risk insights from the RBS living PSA. Note that this balance is important since each element cannot be used on its own to define the performance criteria. Current industry practices provide us with expectations from other utilities. Our living PSA includes historical insights that have been updated regularly since the IPE submittal, including failure data that has been trended since 1988. By merging these two elements combined with expert panel judgement, we can establish performance criteria that account for the past, but also allow for valid statistical changes in the failure data over long time periods in the future. Note that this approach does allow deviations from probabilities that can be derived from the data in the PSA. This is to be expected, since the actual plant-specific data used in the living PSA is subject to change over time. Over time, the plant specific data in the living PSA may converge to the individual probabilities in the Maintenance Rule program, or it may degrade to an unacceptable level. The conditional probability method allows for these changes without compromising the standard for maintenance that supports plant safety.

Safety Significance - CDF Impacts

The CDF impacts of the four affected systems were estimated using Revision 2B and 2C of the RBS PSA models. The effective dates for the Revision 2B model are from about May 1996 to September 1997. This model is based on plant-specific data from 1988 through August 1994. Revision 2C addresses power operation since September 1997 (Refueling Outage No. 7) and includes changes to operator actions, EDG start failure rates, and model changes. The probabilities below are derived from the Maintenance Rule Performance Criteria (conditional probabilities):

RHR: Maint = 2.0E-2 Fail to Start = 2.0E-2 Fail to Run = 1.0E-1
Prob to Start = 0.98 Prob to Run = 0.90 Reliability = 0.882

HPCS: Maint = 1.5E-2 Fail to Start = 1.0E-2 Fail to Run = 3.9E-2
Prob to Start = 0.99 Prob to Run = 0.96 Reliability = 0.95

RCIC: Maint = 3.5E-2 Fail to Start = 2.0E-2 Fail to Run = 1.2E-1
Prob to Start = 0.98 Prob to Run = 0.88 Reliability =0.862

EDG: Maint = 2.5E-2 Fail to Start = 2.0E-2 Fail to Run = 2.7E-2
Prob to Start = 0.98 Prob to Run = 0.97 Reliability = 0.95

Note that Reliability = Probabability to Start * Probability to Run.

RHR: Train A or B Maint = 1.35E-2 Train C Maint = 1.03E-2 Train level Reliability ~ .90

HPCS: Maint = 1.0E-2 Reliability = 0.982

RCIC: Maint = 5.6E-2 Reliability = 0.824

EDG (each train):
Maint = 2.2E-2 Reliability = 0.956

The train level reliability for RHR was estimated to be about 0.90 based on the cutset data for train A. The changes to the Revision 2C model changed the RHR Train level Reliability to 0.996 for trains A and B and 0.993 for train C. The Revision 2C EDG train reliability to decreased to 0.930 due to the use of more realistic failure to start data in the model.

From the Maintenance Rule conditional probability data, estimates of CDF were calculated by adjusting the data in the model to be consistent with the maintenance rule performance criteria. The first set of sensitivity cases uses the Maintenance Rule conditional probability data in the PSA, Rev. 2B.

Sensitivity Case No. 1 - Rev. 2B PSA Model

Data adjusted for:	CDF	CDF Increase Over Baseline
RHR	1.920E-6	1.3%
RHR + HPCS	2.169E-6	14.5%
RHR+HPCS+EDGs	2.115E-6	11.6%
RHR+HPCS+EDGs +RCIC **(Total Impact)**	**2.012E-6**	**6.17%**

Sensitivity Case No. 2 - Rev. 2C PSA Model

Data adjusted for:	CDF	CDF Increase Over Baseline
RHR	1.965E-6	1.03%
RHR+HPCS	2.601E-6	33.7%
RHR+HPCS+EDGs	2.492E-6	28.1%
RHR+HPCS+EDGs+RCIC **(Total Impact)**	**2.323E-6**	**19.43%**

<u>In each case, the increases in CDF are within the guidance of the NEI/EPRI PSA Applications Guide.</u> The baseline CDF for Revision 2B was 1.90E-6 and for Revision 2C it is 1.95E-6. From the PSA Applications Guide, the non-risk significant range is bounded by an increase of about 72% for each revision. The Maintenance Rule RHR data results in increases in each case, when the total impact is determined. The HPCS data provides the largest increases in CDF for both cases. In these models, we increased the failure to start and failure to run rates of the HPCS pump to account for the difference between the Maintenance Rule conditional probability and the PSA values. This means that we concentrated the failures in the HPCS pump rather than spreading them out between the pump, valves, operator actions, and other components.

This is considered to be a conservative approach which will give higher CDFs. Including the EDG data decreases the CDF, as does the RCIC data. The effect with RCIC is very pronounced and is due to 3 (out of the 4 total) failures that occurred in the timeframe between December 1988 and December 1990. This data was used in the PSA and thus the current failure to start rate in the PSA is higher than what is used in the Maintenance Rule conditional probability method.

Specific concerns raised by NRC during the Maintenance Rule Baseline Inspection at RBS involved RHR and HPCS. The RHR run reliability of 90%, from the Maintenance Rule Program, translates into a failure to run probability of 1.0E-1. This failure rate is very high in comparison to the current Revision 2C PSA and should be revised; however, it compares favorably to the Revision 2B PSA model. The HPCS availability of 98.5%, from the Maintenance Rule Program, is represented by a maintenance unavailability of 1.5E-2. This is very close to the 1.0E-2 value used in the PSA and does not make a significant difference. The higher hardware failure rates derived from the Maintenance Rule conditional probability have more of an impact on CDF than the maintenance value. However, as stated previously, all of the increases above are in the non-risk significant range as defined by the PSA Applications Guide.

Conclusions

The conditional probability method provides an appropriate balance between availability and reliability using industry practices, expert panel judgement and PSA insights. Estimates of the core damage frequencies for the applicable systems show increases in CDF that are not risk significant based on the PSA Applications Guide. The RBS Maintenance Rule program establishes goals for system and equipment performance while balancing this with the historical input from the RBS PSA; these two are related but because of their different nature, do not need to have identical values.

Enclosure 2

CONTAINMENT ATMOSPHERE and HYDROGEN MONITORING SYSTEM (552)

Introduction

Due to a scoping change, inadequate performance criteria had existed since July 10, 1996 for the hydrogen monitoring system such that goals were not established in accordance with section (a)(1) of the maintenance rule.

General Discussion

The Hydrogen Monitoring System, System 552, was originally scoped into the Maintenance Rule as a normally running, non risk significant system. The system functions that using for the scoping determination for the rule are shown in Table 1. The majority of the system components as well as the functions, with the exception of the Hydrogen Analyzers, require continuous operation.

In August of 1997 the Maintenance Rule Coordinator was reviewing the system functions and found that one function listed in System 552 was a standby function in that hydrogen monitoring was required in response to a design basis accident. The Maintenance Rule Coordinator suggested a change in performance criteria to the System Engineer. The System Engineer agreed that the Hydrogen Monitoring Subsystems were standby components that were more appropriately monitored on the train level. An availability goal of 95% was chosen based on providing an effective measure of performance and has no other basis. The performance criteria change was proposed to the Expert Panel (EP) and was approved in the second quarter summary update on 8/21/97.

The Maintenance Rule Coordinator immediately conducted a data review to determine the hydrogen monitoring train availability and determined the availability was below the goal level. He placed System 552 on the next EP agenda for evaluation of (a)(1) status. On September 4, 1997 the system engineer presented the system to the EP. The engineer explained (as documented on panel minutes), recent corrective actions had improved availability and no recent failures had occurred since the CA's were taken and that he expected availability to increase above the goal level by the first quarter of 1998. The EP was not satisfied by the current availability of the hydrogen analyzers and discussed whether the current Availability and Reliability performance criteria was appropriate. Based on the confidence of the system engineer that he had resolved the equipment performance problems, the current goals were considered adequate to monitor the system performance and that the MRC would track availability closely. The decision was made to leave the system in (a)(2).

Based on the change to a Standby System, the system engineer reviewed historical equipment performance. Several events were reclassified as Functional Failures as a result of this change, however no MPFFs were identified based on the known inherent design and equipment weaknesses with the installed equipment. During the previous EP reviews of Quarterly

Summary information no functional failures were recorded as the existing monitoring was being evaluated on the system level and the hydrogen monitoring function could be provided by the redundant operable hydrogen monitoring train. During RF7 and immediately after 10/28/97 additional hydrogen monitoring STP failures occurred. As a result of these failures, the expert panel decided to convene a meeting to reevaluate the System 552 hydrogen monitoring subsystem for (a)(1) status. The EP meeting was held October 30, 1997 and the hydrogen monitoring subsystem was made (a)(1) with the following goals:

- 95% availability per train for H2 monitoring subsystem
- < 1 MPFF / train

There have been a number of calibration related failures associated with the CMS Hydrogen Analyzers. Table 2 provides a listing of all Condition Reports issued during 1996 and 1997. For the Division I analyzer, no failures were noted during 1996 and three for 1997. For the Division II analyzer, five failures were noted, two in 1996 and three in 1997.

For the Division I analyzer, the investigation for the first failure (2-4-97) did not identify any specific cause for the out-of-calibration failure, the instrument recalibrated perfectly and it was placed back into service. However, the second failure (4-1-97) did identify that two differential pressure regulators had failed and this could have been the cause of the first failure. These regulators were replaced, the instrument recalibrated, and placed back into service. A successful calibration was performed during July which indicated that the corrective actions taken were successful. However, another failure occurred during the August calibration attempt. The cause of this failure was indeterminate, the analyzer was recalibrated successfully, and placed back into service.

As part of the investigation for CR 97-0424, it was determined through a review of the surveillance procedure that the calibration tolerance was overly conservative, i.e., required a smaller tolerance than what the analyzer was actually capable of providing. Additionally, it was determined that a further penalty was being taken by not taking into consideration the actual percentage of the calibration gas that was used for the calibration span check. Typically, the hydrogen concentration of the calibration gas that is used for determining the calibration span is of a less percentage than the maximum span of 10%, i.e., 9.87% gas is typically used for the span check. This difference in the actual calibration gas used created conservatism for the operation of the analyzer, however it also reduced the possibility of a successful calibration. The corrective actions to CR 97-0424 included increasing the calibration tolerance and incorporating applicable steps into the surveillance procedure to account for the percentage of calibration gas being used during the calibration attempt.

For the Division II analyzer, each of the first three failures were border line failures and would not have been considered failures had the new tolerances developed later in response to CR 97-0424 been issued and incorporated into the surveillance procedure prior to performance.

Based on the repetitive failures experienced on the Division II analyzer, a corrective action was implemented to increase the calibration frequency for this analyzer to monthly rather than on a quarterly basis as required by the current Technical Specification. This was done in an effort to help identify the cause of the failures and to increase the accuracy of the analyzer should post accident operation be required. Other corrective actions were taken to replace defective components within the analyzer and all of the corrective actions combined seem to have increased the reliability of this monitor. For the period between 5-10-97 to the latest failure on 10-31-97, the monitor was tested three times and had successfully met the acceptance criteria specified.

As a result of the investigation for the failure of the Division I analyzer documented on CR 97-1368, on Wednesday, November 12, 1997, both divisions of the CMS Hydrogen Analyzers were recommended to be considered inoperable. This recommendation resulted from an in-depth review of past calibration data for each analyzer. One of the corrective action items that resulted from a previous condition report, CR 97-0424, required that additional data be collected during the performance of the surveillance testing in an effort to help identify the cause of the calibration failures. This additional calibration data revealed that the existing surveillance calibration test procedure for each analyzer was inadequate in requiring that system operating pressures be within vendor established limits, as measured at the upper and lower test tee of the Analyzer Bypass Flow Indicating Controller. Past calibration data indicates that the as-left pressures at these test points for each monitor were outside of the vendor established limits. This condition could have contributed significantly to the inconsistent calibration results.

The corrective actions that will result from this finding include correcting the surveillance test procedures, adding additional steps to record as-found/as-left values for the Analyzer Bypass Flow Indication, increasing the calibration interval for the Division I analyzer to monthly instead of quarterly, and implementing an administrative as-found calibration limit such that if the limit is exceeded, recalibration will be required. It appears based on a review of past calibration results, that each analyzer is drifting slowly downward over time and calibration failures are not as a result of equipment failure. This corrective action will help prevent the analyzer from drifting beyond the operational limit and causing test failures. **One additional corrective action will be issued to investigate and confirm the vendors accuracy statement for the hydrogen analyzers.** This activity alone may yield substantial benefits beyond what has already been accomplished.

Safety Significance

The Hydrogen Analyzers are used during post accident operation where hydrogen generation may occur due to core damage. The Hydrogen Analyzers are automatically started upon receipt of a LOCA signal and the hydrogen concentration that is detected is used in conjunction with the Emergency Operating Procedures to start and stop various hydrogen control devices at RBS. Starting of the hydrogen control devices is necessary to control the hydrogen concentration within the drywell and containment atmosphere in an effort to prevent the accumulation of explosive amounts of hydrogen gas. Should the hydrogen concentration level increase above the explosive limit that could challenge containment integrity, the hydrogen control systems are secured to prevent hydrogen detonation.

The PSA, DELPHI method, and the EP consider the hydrogen monitoring system a non-risk significant system. Actuation of the hydrogen control systems is required by the EOPs during post accident conditions when the minimum detectable hydrogen concentration level is detected, 0.7% H_2, or when reactor water level falls below -162" (TAF) or when reactor water level cannot be determined. Manual sampling is provided as an alternate method of measuring the hydrogen concentration level within the containment and drywell when both of the redundant hydrogen analyzers are inoperable.

Conclusions

Although the Hydrogen Analyzers were not initially scoped into the Maintenance Rule properly, the Corrective Action Program utilized at RBS provided the means to ensure that failures were reported and appropriate corrective actions implemented. RBS utilized vendor literature as well as telephone conversations and site visits by the vendor, review of calibration data, and discussions with other utility personnel, to identify all possible corrective actions or maintenance activities that could be identified to prevent these failures. Not until the data recorded during the most recently performed surveillance tests was evaluated did an additional cause of the failures become apparent. This data was collected based on a corrective action to a previous failure to help identify and to ensure that the cause of these failures was known. RBS has increased the surveillance frequency on both divisions to ensure the success of the corrective actions and minimize the probability of a surveillance test failure.

Based on the above, it is evident that the intent and spirit of the Maintenance Rule was and is being maintained, i.e., when the performance or condition of an SSC does not meet established goals, appropriate corrective action must be taken. Although specific Availability Goals were not established for this non-risk standby subsystem for the interval between July 10, 1996 and August 21, 1997, it is evident that appropriate corrective measures have been implemented to improve reliability and maintain operability of this system. Corrective measures will continue until the established RBS goals for this system are attained.

Table 1

MAINTENANCE RULE SYSTEM FUNCTIONS
CONTAINMENT MONITORING SYSTEM (CMS-552)

1. Provide containment and drywell pressure indication and alarms for normal and post accident operations.

2. Provide containment and drywell atmosphere temperature indication and alarms for normal and post accident operation.

3. Provide suppression pool temperature and level indication and alarms for normal and post accident operation.

4. Provide remote shutdown panel indication for drywell pressure, temperature, and suppression pool level and temperature.

5. Provide post accident hydrogen concentration indication and high alarm for the containment and drywell atmosphere.

TABLE 2

SYSTEM 552 CONDITION REPORTS
ISSUED DURING 1996 AND 1997

CR NUMBER	DIVISION	DATE	COMMENTS
97-0099	1	2/4/97	As-Found - pegged hi - recalibrated
97-0424	1	4/1/97	As-Found - 6.7% - parts replaced and recalibrated
97-1368	1	9/12/97	As-Found - 8.0% - recalibrated
96-1706	2	10/1/96	Based on CR 97-0424, the tolerance changed and this would not have resulted in a failure. **As-Found 9.5% -recalibrated**
96-2104	2	12/26/96	Based on CR 97-0424, the tolerance changed and this would not have resulted in a failure. **As-Found 9.5% -recalibrated**
97-0489 (Admin Closed to CR 96-1706)	2	4/14/97	Based on CR 97-0424, the tolerance changed and this would not have resulted in a failure. **As-Found 9.6%** - recalibrated
Included as part of CR 96-1706 (monthly calibrations)	2	5-10-97	As-Found 9.4%
97-1937	2	10/31/97	As-Found - 9.25

GAITRONICS POWER SUPPLY

Introduction

A power supply for the Gaitronics communication system was not included within the Maintenance Rule scope when the system was added.

General Discussion

The Gaitronics communication subsystem receives non-Class 1E uninterruptible 120V AC power from three inverters; BYS-INV01B, BYS-INV02 and BYS-INV04. When the Gaitronics system was added to the scope of the rule, BYS-INV04 was inadvertently omitted. This inverter serves 44 of the 66 total areas.

Safety Significance

The Gaitronics subsystem is a part of the Communications system (551) which is a non risk significant system. The Gaitronics is included in scope because it can be used by operators to implement EOP actions. Radios and telephones are also in scope as the remaining parts of the Communications system and also used in EOPs. A complete loss of Gaitronics is not significant because radios and telephones could be used instead to implement EOPs actions.

Conclusion

BYS-INV04 was inadvertently omitted from the maintenance rule scope when the system was added. However, since BYS-INV01B and BYS-INV02 have been in scope, only a portion of the power for the Gaitronics subsystem was not scoped in the rule. Radios and telephones may be used instead of Gaitronics to implement EOPs. As a result of this finding, a CR was written and BYS-INV04 has been placed in scope.

PERFORMANCE CRITERIA FOR THE PRIMARY CONTAINMENT INTEGRITY SYSTEM

Introduction

The purpose of the Primary Containment Integrity System (PCI) is to limit offsite radiation exposure to within 10CFR100 requirements. To ensure that these requirements are maintained 10CFR50 Appendix J requires that the allowable leakage from containment be maintained at less that 1.0La. La is defined as the maximum allowable leakage, expressed in % per 24 hours, at the peak accident pressure Pa, expressed in psig. Even though every commercial nuclear power plant must comply with the 1.0La requirement, the actual leakage value associated with La varies from plant to plant.

The RBS Technical Specification Bases defines "the maximum allowable leakage rate for the primary containment (La) is 0.26% by weight of the containment and drywell air per 24 hours at the design basis LOCA maximum peak containment pressure (Pa) of 7.6 psig." The value of 0.26% by weight per 24 hours is equivalent to 110,748 sccm. The Technical Specifications also states that the leakage rate acceptance criteria for Type B and C components is ≤0.6La and the acceptance criteria for the Type A test is ≤0.75La.

General Discussion

Reg Guide 1.160, "Monitoring the Effectiveness of Nuclear Power Plants," indicates that proper plant maintenance is essential for ensuring that the health and safety of the public is maintained. Per Section 1.2, Paragraph 1, the objective of 10CFR50.65 is to require the monitoring of overall effectiveness of licensee maintenance programs to ensure that safety-related and certain nonsafety-related SSCs are capable of performing their intended functions. Therefore, monitoring the effectiveness of maintenance is directly related to ensuring that system functions are maintained.

The function of the PCI System is to limit the offsite radiation exposure to less than the limits stated in 10CFR100. For Maintenance Rule purposes (10CFR50.65), the primary containment performance criteria is not satisfied when there are more than 2 MPFFs in an 18 month period. A "functional failure" exists when the test results for a component causes the leakage from PCI to be in excess of 0.6La. Past functional failure evaluations performed for RF-4 thru 6 applied the 0.6La performance criteria to each valve.

PCI is considered to be a standby system, and its performance is verified via the Surveillance Testing Program. Present functional failure evaluations apply the test results of each valve along with the remaining Type B & C components (including the annulus bypass leakage paths) against the 0.6La performance criteria. That is, the combined leakage rates are determined by employing the latest leakage rate test data available, and are maintained as a running summation. RBS's 10CFR50 Appendix J, Option B program currently requires a cause determination be performed per RBNP-030, "Initiation and Processing of Condition Reports," whenever the administrative limit of a Type B & C component is exceeded. This ensures that the impact of component degradations are evaluated against the system performance criteria which is also consistent with 10CFR50 Appendix J, Option B. Evaluating functional failures in this manner is conservative because the function of the system is not lost at that leakage rate. Assigning a value of 0.6La as a performance criteria ensures that the function of the containment structure is maintained (i.e., limit offsite dose to less than the 10CFR100 limits).

Safety Significance

LOCA dose calculation, G13.18.9.5*051-0A, assumes a containment leakage rate of 0.26%/day for the duration of the event. There are three major contributors to the off-site dose. The first is leakage from containment and secondary containment. The second is the leakage through the PVLCS lines. The third is due to liquid leakage from ESF systems.

A sensitivity study for both LPZ and EAB thyroid dose was performed by varying the overall containment leakage rate, La. The results of the analysis demonstrates that, per the current licensing basis calculation, increasing La by a factor of approximately 5 would not cause 10CFR100 dose limits to be exceeded.

Applying a more restrictive maintenance rule performance criteria will not significantly reduce the risk to the public. Per NUREG-1493, page 1-2, "information on reactor accident risks derived from PRA assessments indicates that the currently allowable containment leakage rates can be increased without significantly affecting accident risk. While availability and reliability of containment integrity are important, the extremely low leakage rates prescribed by current regulations and the testing measures taken to assure these extremely low leakage rates may not be warranted. Reactor accident risk is dominated by low-probability, high-consequence scenarios in which the containment is failed or bypassed. In these types of accidents, there is little benefit derived from a high degree of containment leak tightness".

Conclusion

NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Reg Guide 1.160 allow the use of functional failures as performance criteria for risk significant SSC's and acknowledge the use of functional failures as a performance indicator. In this case, a performance criteria of 0.6La is an appropriate performance criteria, and it provides a conservative indication of the effectiveness of maintenance before the function is lost.

Developing a more restrictive performance criteria (< 0.6La) will not significantly reduce the risk to the public. However, the RBS Expert Panel has assigned an action item to the responsible System Engineer to evaluate the performance criteria for components under the Appendix J Option B Program giving consideration to changing the performance criteria from ≤ 2 MPFFs per 18 months to the following:

≤ 1 administrative failure in any 3 rolling test periods per component

or

the combined leakage rates of Type B & C components is > 0.5La

PERFORMANCE CRITERIA FOR STRUCTURES

Introduction

A question was raised during the Maintenance Rule inspection which indicated that performance criteria for structures would not allow structures to reach (a)(1) status.

General Discussion

Selected structures were placed in scope of the Maintenance Rule as a result of their function to provide environmental protection and structural support of plant systems and components. Additionally, structures limit offsite radiation exposure to within the limits of 10CFR100 requirements.

The performance criteria utilized for the baseline Maintenance Rule structures evaluation was zero unacceptable conditions for each structure as determined by the Civil/Structural Design Engineering Group. Criteria used to establish acceptability was problem specific based upon the engineering element being investigated (concrete, structural steel, baseplates/anchorage, roofing, coatings, et cetera). The criteria includes the design and licensing bases codes and standards employed in the design of the structures. This position is consistent with the guidance provided in NEI 96-03, "Industry Guideline for Monitoring the Condition of Structures at Nuclear Power Plants," Section 3.4. RBS Structural Design Criteria Document Number 200.010, Revision 5 was used for the baseline inspection and remains the primary basis for designing and analyzing structures.

Structures are classified as (a)(1) due to either a present as-found condition resulting in a structure not being able to satisfy all design basis loading requirements, or a structure or structural component in a deteriorated state such that a failure to meet all design basis loading requirements is possible before the next regularly scheduled inspection. Failure to satisfy all design basis requirements constitutes a functional failure.

The Civil/Structural Design Engineering Group's assessment of structures is programmatically infused into the Maintenance Rule monitoring program. Issues discovered and evaluated by structural monitoring, both initial baseline and periodic, are reviewed and addressed by the Maintenance Rule Expert Panel prior to approving the results of the monitoring period. This includes all conclusions reached for (a)(1) and (a)(2) assessment determinations.

Site Procedures PEP-0219, "Reliability Monitoring Program," EDG-PR-0001, "Reliability Monitoring Program," and EDG-CS-003, "Maintenance Rule Structural Monitoring at River Bend Station," define the process for structural monitoring. Instructional enhancements have been recently achieved via issuance of EDG-CS-003 which did not exist during the baseline structures inspection and evaluation. RBS procedures are aligned with NEI 96-03. Performance criteria in place during the initial structural monitoring is the same criteria currently employed in the periodic monitoring program.

Safety Significance

Maintenance of the integrity of structures is essential to ensure the environmental protection and structural support of plant systems and components are preserved for all design basis conditions. The ability of structures to limit offsite radiation exposure must also be preserved.

Conclusion

The River Bend process for monitoring structures has always included appropriate guidance for determining (a)(1) or (a)(2) status. The criteria that existed during the original baseline inspections that would have been utilized to identify an unacceptable condition, is the same criteria that exists in the current procedures. If an unacceptable condition existed during the baseline inspection, the same codes and standards that exist today would have been utilized to evaluate deficiencies.

Plant structures within the scope of the Maintenance Rule have been evaluated and no conditions exist which adversely impact the ability of structures or sub-components from satisfying all design basis requirements both in the present state or the projected condition prior to the next scheduled inspection. It should be noted however, that proactive improvements have been taken as a result of the Maintenance Rule structural monitoring program. Fasteners on the support structure for the transverse incore probes were replaced, building expansion joint material at seismic joints has been replaced, changes to roof drainage elements are underway and repetitive task preventive maintenance work instructions generated to improve the condition of plant coatings in the Drywell. Although proactive improvement measures have been undertaken as a result of the Maintenance Rule monitoring program, no conditions have been discovered warranting classification as (a)(1).

CONTAINMENT ISOLATION VALVES (CIV) CAUSE DETERMINATION

Introduction

During RBS Maintenance Rule Baseline Inspection 97-09, five Containment Isolation Valves (CIV) were identified with apparent inadequate cause determinations. The five CIVs are C11-VF122, DFR-V180, IAS-V80, SSR-V706 and SVV-V31.

General Discussion

At the time of the above mentioned failures, there were no programmatic requirements to perform cause determinations for Local Leak Rate Test (LLRT) valve failures. Specifically, ADM-0050, "Primary Containment Leakage Testing Program" Revision 4, paragraph 5.19.2, stated that a condition report (CR) should be initiated only if there is evidence of a trend of degraded performance of the valve. Currently, the LLRT program requires that a CR be written for every LLRT result that exceeds the administrative limit.

For 4 of the subject valves identified above, LLRT results during the period monitored (RF-5 and RF-6) did not indicate a trend of degraded performance; therefore, no CR was initiated. However, for valve DFR-V180, consecutive LLRT failures occurred during RF-5 and RF-6 testing. As a result, CR 96-0285 was initiated to document the trend of LLRT failures. For both test failures for valve DFR-V180, internal debris was determined to be the cause.

Subsequently, as part of the Maintenance Rule Program implementation, an Initial Assessment Report was developed which evaluated, in part, the LLRT failures for the subject valves that had occurred during RF-5 and RF-6 for classification in accordance with the maintenance rule. For each of the subject valves, the Initial Assessment Report classified each failure as a functional failure (FF) due to debris in the valve. The corrective actions performed included cleaning the valve internals, inspection and replacement of worn parts, and checking the valve seats. Subsequent to the actions, each valve passed the as-left LLRT.

Based on a recent review of the Initial Performance Assessment Report cause statements and classifications, it is apparent that all of the subject valve failures should have been classified as MPFFs, since the initial failure cause, debris in the valve, was preventable by maintenance such as periodic blowdowns, flushes, and filtration. Nevertheless, corrective actions were performed for some of these systems in the 1995-1997 time period that have reduced the number of LLRT failures for all five valves and changed the Maintenance Rule classification for two valves. These are summarized below.

A summary of the corrective actions and results are as follows:

SVV-V31: The improved performance is due to the extensive flushes of the SVV system performed in accordance with CR 96-0048 during RF-6, and the completion of MR 96-0073 during RF-7 that replaced carbon steel piping in the SVV system with stainless steel, eliminating the primary debris source. Based upon the recognition that a design problem was the primary cause for the rust particulates in the system, this failure was adequately classified as a functional failure (FF).

DFR-V180: This valve had LLRT failures in RF-5 and RF-6 but passed in RF-7. The improved performance is most likely the result of CR 95-0689 corrective actions which identified incorrect valve internals (spring and disc) originally installed by the vendor. The valve internals were replaced in RF-6.

The classification was recently revised to MPFF since the internal debris could not be eliminated as a contributing cause.

IAS-V80: The improved performance was the result of MR 93-0005 which installed a new "heatless" air dryer, new centrifugal compressors, and flushed the system which virtually eliminated the problem of moisture intrusion into the IAS system and greatly reduced the generation and entrainment of rust particles. Based upon the recognition that a design problem was the primary cause for the rust particulate in the system, this failure was adequately classified as a FF.

C11-VF122: The LLRT failure in RF-6 was the first failure of this component since 1985. No repeat failures have occurred. Therefore, no system specific corrective action is planned as a result of this isolated failure. The classification was recently revised to MPFF.

SSR-V706: The rework performed in RF-5 (replacing the disc and spring) apparently contributed to the improved sealing capability of the valve as indicated by the satisfactory LLRT results in RF-6 and RF-7. The classification was recently revised to MPFF since rework of the disc and spring were needed to improve valve performance.

Safety Significance

Although there are instances where initial maintenance rule classifications were incorrect, there is no safety significance associated with this event. Corrective actions were initiated as a result of corrective action program requirements which has resulted in improved valve performance.

Conclusion

CRs were not initiated for the initial subject LLRT failures; nevertheless, a cause determination was performed during the Initial Performance Assessment of the PCI system based upon the maintenance document descriptions of each valves' as-found condition. Upon further review of the assessment report, it has been determined that the initial Maintenance Rule classifications for each valve was in error. Regardless, corrective actions were implemented that provided significant LLRT performance improvements.